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November 14, 2007

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Subject: Licensee Event Report 50-458 / 07-005-00
River Bend Station – Unit 1
Docket No. 50-458
License No. NPF-47

File Nos. G9.5, G9.25.1.3

RBG-46760
RBF1-07-0208

Ladies and Gentlemen:

In accordance with 10CFR50.73, enclosed is the subject Licensee Event Report. This document contains no commitments.

Sincerely,

David N. Lorfing
Manager – Licensing

DNL/dhw
Enclosure

IE22

NRR

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NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Unplanned Reactor Scram During Surveillance Testing Due to Damaged Terminal Board

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
09	26	2007	2007	- 005 -	00	11	14	2007	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
David N. Lorfing, Manager – Licensing	225-381-4157

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On September 26, 2007, at 10:42 pm CDT, an unplanned automatic reactor scram occurred while the plant was operating at 100 percent power. At the time of the event, scheduled surveillance testing was in progress for a functional test of the average power range monitor (APRM) channel "A". Part of the test procedure involved the actuation of the Division 1 reactor protection system (RPS) trip circuitry. When this action was taken, 36 reactor control rods ("Group 2" rods) unexpectedly inserted into the core. As the reactor operator was taking actions to respond to this condition, an automatic reactor scram was generated by a low reactor water level (Level 3) signal. This event is being reported in accordance with 10CFR50.73(a)(2)(iv) as an automatic actuation of the reactor protection system. The investigation found that a terminal block and wiring had been damaged by overheating due to a loose terminal screw, which had caused a loss of power to the scram valve pilot solenoids on the Group 2 rods. This loss of power was not apparent to the operators, as it occurred in a part of the circuit downstream of the power status lights. The damaged components were repaired, and similar circuits were inspected for loose terminals.

REPORTED CONDITION

On September 26, 2007, at 10:42 pm, an unplanned automatic reactor scram occurred while the plant was operating at 100 percent power. At the time of the event, scheduled surveillance testing was in progress for a functional test of the average power range monitor (APRM) channel "A". Part of the test procedure involved the actuation of the Division 1 reactor protection system (RPS) trip circuitry. When this action was taken, 36 reactor control rods ("Group 2" rods) unexpectedly inserted into the core. As the reactor operator was taking actions to respond to this condition, an automatic reactor scram was generated by a low reactor water level (Level 3) signal. This event is being reported in accordance with 10CFR50.73(a)(2)(iv) as an automatic actuation of the reactor protection system (JC).

The low reactor water level signal resulted from the decrease in reactor power that followed the insertion of the Group 2 control rods. The Level 3 condition also caused an actuation of the containment isolation system, as designed. The isolation valves that respond to a Level 3 signal were already closed. No reactor safety relief valves actuated in response to this event. Following the initial transient, the operators promptly stabilized reactor pressure and water level.

INVESTIGATION and CAUSAL ANALYSIS

At the time of this event, all emergency core cooling systems were in their normal standby configuration. The Division 1 diesel generator (DG) was running for a scheduled monthly surveillance test, and the Division 2 and 3 DGs were in standby.

The APRM surveillance test contains steps where an actuation signal of the Division 1 reactor protection system is intentionally generated in order to test the circuitry. This "half-scram" actuation does not cause any actual control rod motion, as both divisions of the RPS must be tripped to accomplish a reactor scram. In order to verify that the RPS system is properly aligned for the test and that no trip signals are already active, the technician is required by the procedure to verify that status lights for the individual RPS channels are energized. This step was properly performed, and the half-scram signal was subsequently actuated. At this point, the Group 2 control rods inserted.

An inspection of components in the affected circuits was performed to verify electrical continuity and proper operation. Engineering and maintenance personnel found that wiring and a terminal board in an RPS pilot scram solenoid circuit had sustained severe thermal damage. This failure had interrupted power to the Division 2 coils on the Group 2 pilot scram solenoid valves, in effect causing an undetected Division 2 half-scram signal for the Group 2 control rods. When the surveillance test inserted the half-scram signal in Division 1, the logic for the Group 2 control rods was completed, and the rods inserted as designed. The circuit failure was downstream of the RPS status lights on the reactor control panel, such that the pre-existing condition of the Group 2 rods could not have been detected by the operator.

A detailed examination of the components determined that the most likely cause of the thermal damage on the terminal board was a loose screw connection on one of the attached wiring lugs. No history of maintenance or testing could be found that might have required the wire to be lifted and re-terminated. It appears likely that the terminal screw had not been sufficiently tightened during plant construction. The thermal damage showed the characteristics of long-term overheating, rather than sudden arcing.

The damaged wiring and terminal board were repaired to restore power to the Group 2 control rod pilot scram solenoids.

CORRECTIVE ACTION TO PREVENT RECURRENCE

To bound the extent of this problem, the RPS pilot scram solenoid circuits were evaluated to determine what portions of the circuit could mask a half-scram. In general, this is generally any point downstream of the RPS status lights. The following actions were taken to address this weakness:

- screws in the terminal boards in the same circuit location were physically verified to be tight, and thermographic readings taken on these terminal boards.
- resistance and voltage measurements were taken on the failed circuit and comparisons made with a known normal circuit to verify that no downstream problem had caused the terminal board overheating.
- thermographic readings were taken on the RPS Group 2 pilot solenoids to confirm that no damage had happened to the solenoids involved.

Plant modifications are being developed to reduce the vulnerability to similar "hidden" conditions of de-energized scram pilot valves. This action is being tracked in the station's corrective action program.

PREVIOUS EVENT EVALUATION

No previous reactor scrams occurring at River Bend Station in the last ten years have been caused by a similar sequence of events.

SAFETY SIGNIFICANCE

The insertion of Group 2 control rods caused a power reduction, which in turn, caused a decrease in steam production which depressed reactor water level. Reactor water level reached the Level 3 scram setpoint approximately six seconds after the Group 2 control rods inserted. A review of the core responses during the event confirmed that neither reactor power nor pressure increased above their initial values. Reactor water level did not decrease to the actuation setpoint for the emergency core cooling systems.

Since Group 2 control rods are evenly distributed throughout the core, the partial scram did not create any significant asymmetry in power distribution. An evaluation of the Group 2 control rod insertion using a core modeling code determined that the core average power was reduced to less than 40 percent. Increases in relative radial, node, and pin powers were significantly smaller than the reduction in core average power, resulting in no challenge to fuel limits. Therefore, the thermal limits and fuel integrity were not challenged by Group 2 control rod insertion.

The reactor response to the event was as expected, and no fission product barriers were challenged. This event was of minimal safety significance to the health and safety of the public.